APPENDIX

PROBABILISTIC RISK ASSESSMENT PROGRAM ELEMENTS

PART I. NRC RESEARCH PROGRAMS IN PROBABILISTIC RISK ASSESSMENT

The international cooperative research effort in probabilistic risk assessment (PRA) has been divided into four general areas of research: (1) Methods Development, (2) Analysis of Operating Events, (3) Development of Advanced PC-Based PRA Software, and (4) Regulatory Applications of PRA. The activities planned in each of these areas are broadly described in the following sections.

1. Methods Development

It is generally recognized that the broad application of PRA to support regulatory decision-making requires methods improvements in a number of risk-significant areas. Among the areas needing improvement are treatment of fire risk, equipment aging, human reliability, and digital systems reliability and risk. NRC programs in these areas are as follows:

a. Fire Risk

The overall purpose of the fire risk research program is to provide technical information in support of the NRC's Risk-Informed Regulation Implementation Plan (RIRIP). In particular, the program will develop fire PRA methods, tools, data, results, and insights needed by the Agency to perform risk-informed decision making.

The fire risk program includes activities that 1) improve qualitative and quantitative understanding of the risk contribution due to fires in operating nuclear power plants (NPPs) and other facilities regulated by the NRC; 2) support ongoing or anticipated fire protection activities in the NRC program offices, including the development of risk-informed, performance-based approaches to fire protection for operating NPPs; and 3) evaluate current fire PRA methods and tools and develop improved tools as needed to support the preceding objectives.

Previous work has led to the development of improved methods, tools, and data in a number of areas, including circuit analysis, fire detection and suppression analysis, and uncertainty analysis; and to the development of fire PRA insights from reviews of past significant fire events. Ongoing work includes efforts to develop comprehensive, state-of-the-art guidance for the conduct of fire PRA and gain insights from plant-specific application; develop (in cooperation with a number of international organizations) an improved understanding of the uncertainties and limitations in current fire models; support ongoing fire-related regulatory efforts (e.g., the NRC's fire protection Significance Determination

Process and associated circuits inspections); and support development of the American Nuclear Society full power fire risk standard.

b. Equipment Aging

The objective of this research effort is to assess the feasibility of using reliability-based physics models to incorporate the effects of aging into an integrated probabilistic risk assessment. Earlier work in this area assessed the feasibility of using this technique for the aging of piping. This work was published in NUREG/CR-5632 in 2001. Additional work in this area is the application of this technique to assessing the effect of aging on the failure of in-containment instrumentation and control cables during a loss of coolant accident. A report will be published in 2004 describing a method of assessing the probability of failure of these cables as a function of their age, and the inservice dose rate and temperature the cables are exposed to, with some numerical examples. Additional work will be dependent on obtaining the cooperation of a licensee to provide data on cable insulation materials and the environment of cables.

c. Human Reliability

The general objectives of the human reliability analysis (HRA) research are to: 1) develop improved human reliability analysis (HRA) methods, tools (including guidance), and data needed to support NRC regulatory activities, including the broad implementation of risk-informed regulation; and 2) develop HRA insights to support the development of technical bases for addressing identified or potential safety issues.

Previous work has led to the development of ATHEANA, an improved method for HRA that focuses on the identification of error forcing contexts that increase the likelihood of human errors; the application of ATHEANA in the assessment of pressurized thermal shock (PTS) risk in support of efforts to re-examine the technical basis for 10 CFR 50.61, the PTS rule; and the development of an improved method for HRA quantification that explicitly treats uncertainties. Current work includes the continual use of ATHEANA in PRA applications (e.g., the fire requantification and steam generator tube rupture); the development of an improved method for HRA quantification that includes the use of evidence from a variety of sources; the development of a repository for human event reliability analysis (HERA); and the development of HRA guidance, i.e., an HRA Good Practices document, to support the use of the American Society of Mechanical Engineers (ASME) PRA standard.

d. Digital Systems Reliability and Risk

The increased use of digital instrumentation and control systems in nuclear power plants is introducing some unique reliability and risk issues. This project will be focused on providing methods for more quantitative, probabilistic assessments of digital systems reliability and their impact on overall plant risk, including hardware and software reliability and human-system interface issues. The staff is currently focusing on Failure Mode and Effect Analysis (FMEA) in support of developing

reliability models of digital systems. The potential goals are finding a better definition of the reliability problems of digital systems and a better process of applying FMEA to digital systems. The future work is expected to be in the areas of software reliability and the failure rate data development.

2. Analysis of Operating Events

a. ASP Program

The Accident Sequence Precursor (ASP) Program was established by the NRC in 1979 in response to the Risk Assessment Review Group report (see NUREG/CR-0400, September 1978). The primary objective of the ASP Program is to systematically evaluate U.S. nuclear plant operating experience to identify, document, and rank operating events most likely to lead to inadequate core cooling and severe core damage (precursors), if additional failures had occurred.

The other objectives of the ASP Program are:

- To categorize the precursors by their plant-specific and generic implications,
- To support performance measures contained in the Agency's annual Performance and Accountability Report to Congress,
- To provide a measure for trending nuclear plant core damage risk, and
- To provide a partial check on probabilistic risk assessment (PRA)-predicted dominant core damage scenarios.

Events and conditions from licensee event reports, inspection reports, and special requests from NRC staff are reviewed for potential precursors. These potential precursors are analyzed, and a conditional core damage probability (CCDP) is calculated by mapping failures observed during the event onto accident sequences in risk models. An event with a CCDP or a condition with a change in core damage probability greater than or equal to 1×10^{-6} is considered a precursor in the ASP Program.

Plant-specific and generic insights and lessons learned from the ASP program, and other issues of interest that were encountered during the precursor analysis of operating experience (e.g., projection of unanticipated accident scenarios, risk exposure from precursors, and adequacy/availability of risk mitigation measures) are currently being exchanged in annual meetings with Organization for Economic Cooperation and Development (OECD) member countries.

b. SPAR Model Development Program

The Standardized Plant Analysis Risk (SPAR) models are the analysis tool used by staff analysts in many regulatory activities, including the ASP Program. The current set of SPAR models includes PRA models for internal initiating events during full power operation for each operating plant in the U.S. In addition, generic models for low-power and shutdown operations, and Level 2/large early release frequency (LERF) analysis are being developed for several plant

categories. Currently, plant specific SPAR models are available only to NRC and licensees.

c. Reactor Performance Data Collection Program and Industry Trends Program

The objectives of these programs are to:

- Collect industry data and produce industry trends for initiating events, common-cause failures, system and component reliabilities, and fire events.
- Establish thresholds for the associated industry trends.
- Develop integrated industry indicators and thresholds for the above.
- Produce parameter estimates for use in the SPAR models and other risk analyses for initiating events, components, and common-cause failures.

The NRC is currently developing a new approach for industry trends. The proposed Baseline Risk Indicator for Initiating Events (BRIIE) uses industry data available from current NRC programs, and is closely tied to risk, e.g., core damage frequency. The BRIIE uses a risk-significant subset of initiating events along with appropriate risk weights obtained from the various plant PRAs.

d. Development of Risk Based Performance Indicators

The NRC is developing a mitigating systems performance index (MSPI) to monitor the performance of six systems based on their ability to perform risk-significant functions. The index comprises two elements - system unavailability and system reliability. Plant-specific PRA models are used to calculate the contribution of component failures and maintenance unavailability to the index, which approximates the change in core damage frequency. The NRC is currently evaluating several technical issues arising from the pilot plant program and is also investigating the feasibility of implementing the MSPI as part of the Agency's Reactor Oversight Process.

3. Development of PC-Based PRA Software

The NRC has developed and maintains the SAPHIRE (Systems Analysis Programs for Hands-on Analysis Integrated Reliability Evaluations) PRA computer code. SAPHIRE offers a state-of-the-art capability for assessing the risk associated with any complex system or facility. In particular SAPHIRE can be used to assess the risk associated with nuclear power plants in terms of core damage frequency (Level 1 PRA) and containment performance and radioactive releases (Level 2 PRA). SAPHIRE includes a Graphical Evaluation Module (GEM), a separate subroutine that provides a simplified user interface for performing analysis using SPAR models, discussed above.

Both the continual advancement of the state-of-the-art in the use of computers and the continual expansion of the use of risk-information in the NRC's decision-making, necessitate continual maintenance and improvement of SAPHIRE.

It is expected that this program will continue to provide software maintenance and user support and expand SAPHIRE capabilities by decreasing size limitations (on the

number of basic events, fault trees, sequences, end states, etc. handled by SAPHIRE), speeding up cutset generation and data analysis using multiple processors, adding work group project integration capability, and creating a web-page type user interface with the goal of reducing complexity without losing SAPHIRE's functionality. Furthermore, SAPHIRE's documentation will be revised by issuing a new report for the Windows Versions 6 and 7. Finally, a SAPHIRE interface is being developed to be used in the Reactor Oversight Process.

4. Regulatory Applications of PRA

   a. Changes to Reactor Regulations

   NRC has been actively pursuing the increased use of PRA methods, models, and insights to support regulatory decisions. Among the active programs are those which use PRA results to identify changes needed in reactor safety requirements. There are currently two regulations 10 CFR 50.44 "Standards for Combustible Gas Control Systems in Light-Water-Cooled Power Plants" and 10 CFR 50.46 "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled Power Plants" that the staff is revising based on current risk information and research results. In September 2003, NRC concluded rulemaking on 50.44 by issuing a risk-informed revision to 50.44 which among other changes, eliminated the current requirements for hydrogen recombiners. Proposals are under consideration for risk-informing 50.46.

   b. Regulatory Guidance on PRA

   The NRC staff has developed a draft regulatory guide (RG) that provides guidance to licensees on how to use PRA standards and industry peer review programs to demonstrate that the risk input to a risk-informed decision is technically defensible. This new RG will be accompanied by a Standard Review Plan (SRP) chapter. The main body of the RG provides guidance on the use of PRA standards and industry guidance by licensees to determine the level of confidence that can be afforded PRA insights/results in support of decision-making. The staff's endorsement of the standards and industry program will be the appendices to this RG. Specifically, Appendices A and B include the staff's position on the American Society of Mechanical Engineers (ASME) PRA standard and the Nuclear Energy Institute (NEI) peer review process respectively both addressing full-power, internal events, excluding internal fire, Level 1 and limited Level 2 (LERF) PRA. As the American Nuclear Society (ANS) PRA standards are issued on external hazards, low power and shutdown and internal fires, additional appendices will be added to the regulatory guide.

   The draft RG was issued in November 2002 for public review and comment. A RG for trial use was issued for pilot applications in February 2004. Pilot applications include different allowed outage time (AOT) for technical specifications changes and 10 CFR 50.69.

c. Risk of Dry Cask Fuel Storage

NRC is performing a pilot PRA of a spent fuel dry cask storage system, the Holtec International HI-STORM 100. This cask is being studied at a specific Boiling Water Reactor (BWR) site where the operations can be observed and modeled. (Although developed for a specific cask at a specific site, the analytical models developed for this preliminary study can be modified and applied to other dry cask systems at other reactor sites.) During its service life, the cask has three operational modes - handling in the reactor building, transfer to the storage pad, and storage for 20 years. In each of these modes, accidents that could result in mechanical and thermal challenges to the cask and that have the potential to cause the release of radioactive material, are postulated. Available data are used to estimate accident frequencies. Engineering analyses are used to determine the stresses that would be imposed by the postulated events. The postulated events include drop accidents during handling in the reactor building and transfer to the storage pad. During the storage phase of 20 years on the storage pad, the postulated events include, but are not limited to, tornadoes, tornado generated missiles, earthquakes, floods, meteorites, and gas line explosions. Fracture mechanics and other engineering disciplines are used to determine the probability of a cask failing when subjected to postulated accident conditions.

The preliminary results of the PRA suggest that the risk to the public of the HI-STORM cask at the BWR plant is very low compared to the risk of accidents involving the core of operating nuclear power plants. Accidents and hazards caused by natural phenomena like seismic, high winds, floods, etc., that have a high conditional probability of failing the cask have a very low frequency. Furthermore, the consequences of the postulated accidents that can fracture the cask and the fuel are low because the energy driving the radionuclides from the fuel pellets is low and the inventory of radionuclides in the fuel pellets is relatively low compared to the reactor inventory. Accordingly, the risk, defined as the sum of the products of the accident frequencies and consequences, is very low.

d. Development of Risk Guidelines for Nuclear Materials and Waste Applications

The NRC Commissioners have approved the staff's plans to continue advancements in risk-informing activities in the nuclear materials and waste arenas as a means of improving the Agency's focus on safety, effectiveness, and efficiency, and in reducing unnecessary regulatory burden. As work is completed in the risk informing activities in the nuclear materials and waste arenas, the information will be shared.

APPENDIX A

PART II. KINS RESEARCH PROGRAMS IN PROBABILISTIC SAFETY ASSESSMENT (PSA)

International cooperative research efforts in PSA have been divided into three general areas of research: (1) Methodology developments, (2) Development of fundamental risk-informed application framework, and (3) Applications of risk information. The activities planned in each of these areas are broadly described in the following sections.

1. Methodology Development

It is generally understood that PSA application to the regulatory decision-making process necessitates modeling improvements and standardization in a number of risk-significant areas. The improvement in terms of PSA methodology is needed in many areas such as assessment for non-operational modes, human performance, etc. The long-term research projects by KINS and KAERI have been in progress in these areas as follows:

a. Low Power/Shutdown PSA

The specific objectives of this project are to develop PSA methodologies for various plant operational modes (POSs) and to evaluate the risk impacts in nuclear power plants during these POSs. The work scope of PSA is currently limited to internal events. How to improve the plant's safety was already proposed as a part of the preliminary conclusion. The project will be continued with ongoing work for methodology standardization.

b. Human Performance and Reliability Analysis

The human being is a critical factor to the safety of NPPs. Due to the lack of empirical data and theoretical background, however, there are high uncertainties in assessing the human factors. For the evaluation and the enhancement of human performance in a reasonable way, current activities are focused on the development of a human performance database and enhanced framework of human error analysis. Specific research topics of the project are as follows:

- Development of human performance database (DB): Human performance data have been collected generally from simulator records and operating experience. With this work some input data on human reliability analysis (HRA) have been generated, such as performance time (including event diagnosis time and step execution time), task types, error types, major performance shaping factors, and so on. The human performance DB would be used as a technical bases for HRA and human factor research.

- Development of a standard method for HRA: The staff is currently focusing on standard HRA methods and procedures in support of developing risk-informed regulatory applications.

- Other Research Areas: Research on the development of task complexity measures, and an analysis method for error of commission are also addressed.

These projects would increase the confidence of HRA, and enhance the human performance of nuclear power plants.

c. Reliability Analysis on Digital Instrumentation and Control Systems

The increased use of digital instrumentation and control systems (I&C) in operating plants raises some unique reliability and risk issues. In Korea, digital I&C will be introduced into the design of the new plants. Therefore, there is a need to develop a methodology to assess the safety and reliability of digital I&C systems in nuclear power plants. Previous work has led to the development of software verification and validation using qualitative method. Current work includes the continued development of quantitative and probabilistic techniques applicable to digital I&C, including software reliability and human-system interface issues.

Through this project and the insights from previous experiences, some critical factors of system unavailability, such as the coverage of a fault tolerant mechanism, the model of common cause failures, and the software failure probability were identified. The effect of each factor through sensitivity studies was examined. The results of sensitivity studies show that unreasonable assumptions on these factors would severely distort the analytical results. Such results will be used as the basis for future research projects.

2. Development of Fundamental Risk-informed Application Framework

In order to resolve concerns related to risk-informed applications and reaching consensus with the Utility, it is undoubtedly necessary to develop a fundamental scheme and techniques for the regulatory body. Korean programs in these areas are as follows:

a. PSA Standardization

The objective of this research effort is to improve regulatory decision making by providing assurance on the technical adequacy of current PSAs, which are performed for licensing purposes or diverse risk-informed applications. Ongoing work includes efforts to (1) assess a quality grade of dedicated existing PSA, (2) develop standardized modeling and a procedure for PSA, and (3) perform a detailed evaluation on special topics for enhancement needed, such as common cause failure analysis and HRA.

b. Development of a Regulatory Audit PSA Model

The analysis tool used by the KINS staff analysts in many regulatory matters is currently developed. The key in preparing the analysis tool is to develop audit PSA models for each of the operating plants in Korea. Through the cooperation

between KINS and KAERI, the work has actively progressed. Currently the focus is on the development of a pilot model for internal initiating events during full power operation.

c. Reliability Database (DB)

A plant-specific PSA requires various sets of accurate reliability data. Earlier work in this area was to set up a Korean plant-specific reliability DB. Three areas of activities were initiated: (1) the development of software to simplify the reliability data collecting process at the shop level. This consisted of an automated shift supervisor's logging system related to the test and maintenance of safety related systems and components in nuclear power plants, (2) the simplification of the centralization process of reliability DB stored at each site. The program structure of the Internet home page was also developed and tested, (3) the development of a data structure for each module to facilitate a reliability DB for the operating plants. Sample data collection and analysis is now underway.

d. PSA Software

The KAERI has developed a computer code package called KIRAP and CONPAS for Level 1 and Level 2 PSA analyses. Efforts to improve its capability and user interface continues. Current work includes activities that include (1) the development and enhancement of a fast cut set generation algorithm, (2) support of user-friendly interfaces, and (3) the development of interface software for PSA DB preparation, parameter updating, etc.

e. Risk Monitoring

By the use of a risk monitor, the operator or maintenance scheduler can monitor the risk status of the operating plant and thus avoid high peak risk by rearranging maintenance work schedules as early as possible during maintenance planning. The previous work in this area was to convert effectively the existing traditional PSA models to risk monitoring ones. The implementation guidance for risk monitoring for each operating plant is currently being prepared by the KINS staff.

3. Application of Risk Information

a. Regulatory Inspections

In Korea, the regulatory authority conducts periodic regulatory inspections during each refueling outage for selected systems, structures, and components (SSCs). The activity on the development of a risk-based regulatory inspection program has been done, where inspection activities are graded mainly according to their risk significance of SSCs and operator actions. The objective of the program is to improve the efficiency in regulation and plant safety. The program has been developed based on plant-specific level 1 PSA results and operating experiences, and focuses on key inspection items categorized by risk significance. The

program also includes inspection insights and inspection checklists that have been derived from domestic and worldwide operating experiences.

Direct implementation of the program needs firm and steady efforts, such as the revision of legal requirements for regulatory inspection, and requires much more consensus on its effectiveness with the KINS staff.

b. Analysis of Operating Events

Although there is no Accident Sequence Precursor (ASP) program in Korea yet, it is a common understanding that the use of risk information regarding operating events is greatly beneficial. Preliminary study for an ASP-concept application to selected operating events was conducted. Future work to develop the ASP program in Korea is expected.

INTELLECTUAL PROPERTY ADDENDUM

Pursuant to Article V of this Agreement:

The Parties shall ensure adequate and effective protection of intellectual property created or furnished under this Agreement and relevant implementing arrangements. The Parties agree to notify one another in a timely fashion of any inventions or copyrighted works arising under this Agreement and to seek protection for such intellectual property in a timely fashion. Rights to such intellectual property shall be allocated as provided in this Addendum.

I. SCOPE

A. This Addendum is applicable to all cooperative activities undertaken pursuant to this Agreement, except as otherwise specifically agreed by the Parties or their designees.

B. For purposes of this Agreement, "intellectual property" shall have the meaning found in Article 2 of the Convention Establishing the World Intellectual Property Organization, done at Stockholm, July 14, 1967; viz., "'intellectual property' shall include the rights relating to:

- literary, artistic and scientific works,
- performances of artists, phonograms, and broadcasts,
- inventions in all fields of human endeavor,
- scientific discoveries,
- industrial designs,
- trademarks, service marks, and commercial names and designations,
- protection against unfair competition, and all other rights resulting from intellectual activity in the industrial, scientific, literary or artistic fields."

C. This Addendum addresses the allocation of rights, interests, and royalties between the Parties. Each Party shall ensure that the other Party can obtain rights to intellectual property allocated in accordance with the Addendum by obtaining those rights from its own participants through contracts or other legal means, if necessary. This Addendum does not otherwise alter or prejudice the allocation between a Party and its nationals, which shall be determined by that Party's laws and practices.

D. Disputes concerning intellectual property arising under this Agreement should be resolved through discussions between the concerned participating institutions or, if necessary, the Parties or their designees. Upon mutual agreement of the Parties, a dispute shall be submitted to an arbitral tribunal for binding arbitration in accordance with the applicable rules of international law. Unless the Parties or their designees agree otherwise in writing, the arbitration rules of the United Nations Commission on International Trade Law (UNCITRAL) shall govern.

E. Termination or expiration of this Agreement shall not affect rights or obligations under this Addendum.

II. ALLOCATION OF RIGHTS

A. Each Party shall be entitled to a non-exclusive, irrevocable, royalty-free license in all countries to translate, reproduce, and publicly distribute scientific and technical journal articles, reports, and books directly arising from cooperation under this Agreement. All publicly distributed copies of copyrighted work prepared under this provision shall indicate the names of the authors of the work unless an author explicitly declines to be named.

B. Rights to all forms of intellectual property, other than those rights described in Section II. A. of this Addendum, shall be allocated as follows:

1. Visiting researchers, for example, scientists visiting primarily in furtherance of their education, shall receive intellectual property rights under the policies of the host institution. In addition, each visiting researcher named as an inventor shall be entitled to share in a portion of any royalties earned by the host institution from the licensing of such intellectual property.

2. (a) For intellectual property created during joint research, for example, when the Parties, participating institutions, or participating personnel have agreed in advance on the scope of work, each Party shall be entitled to obtain all rights and interests in its own country. The Party in whose country the invention was made shall have first option to acquire all rights and interests in third countries. If research is not designated as "joint research", rights to intellectual property arising from the research will be allocated in accordance with Section II.B.1. of this Addendum. In addition, each person named as an inventor shall be entitled to share in a portion of any royalties earned by either institution from the licensing of the property.

   (b) Notwithstanding Section II.B.2.(a) of this Addendum, if a type of intellectual property is available under the laws of one Party but not the other Party, the Party whose laws provide for this type of protection shall be entitled to all rights and interests worldwide. Persons named as inventors of the property shall nonetheless be entitled to royalties as provided in Section II.B.2.(a) of this Addendum.